1. Introduction
A new reactor burnup strategy CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy producing) was proposed, where shapes of neutron flux, nuclide densities and power density distributions remain constant but move to an axial direction.

Application of this burnup strategy to neutron rich fast reactors makes excellent performances. Only natural or depleted uranium is required for the replaced fresh fuels. About 40 % of natural or depleted uranium undergoes fission. But presently, there are no data for material integrity under a condition of 40% burnup.

We try to solve this program by the following method. Fuels burning in progress are removed from core, and recladding is performed. During this process, FP gas is removed from these fuels. Then, they are charged in the previous position of core again. Finally, 40% burnup is attained with maintaining material integrity.

2. Description of the actual work
We have analyzed a fast reactor using CANDLE burnup method. The fuel is natural uranium nitride. Fig.1 shows fuel management for this analysis. Fuel elements “3a” and “6a” in Fig.1 are prepared beforehand for initial fuel exchange. These fuels (3a and 6a) contain similar composition as object fuels “4” and “7”. Then, fuels “4” and “7” are removed from core and fuels “3a” and “6a” are inserted in the corresponding positions, respectively. At the same time, fuel “1” removed from core and fuel “9” consisting of natural uranium nitride was added. It is repeated as shown in Fig.1. Operation time for 1 cycle is about 1600 days and shut-down time is 100 days.

3. Result
The limit of fast neutron fluence is $5.0 \times 10^{23} [/\text{cm}^2]$. Fast neutron fluence distributions at BOC for both with and without recladding are shown in Fig.2, where fuel blocks are removed and recladded. Positions of removed fuel blocks are 165 ~ 170cm and 225 ~ 240cm. In this analysis, a height of one fuel block is 15cm. In calculation, mesh size is 5cm in fuel region, so 3 meshes are regarded as one fuel block. By performing recladding neutron fluence becomes small and less than maximum permissible value.

4. Conclusion
For maintaining material integrity, recladding was adopted in CANDLE fast reactor and exposed neutron fluence can be kept under the limit, even though the 41% burnup of discharged fuel is attained. The effects on reactor physics characteristics of recladding appeared small. This method can keep characteristics of CANDLE burnup strategy.

Reference
Analysis of Recladding in CANDLE Reactor

Akito NAGATA
Tokyo Institute of Technology
2-12-1 N1-17, Ookayama, Meguro-ku,
Tokyo
Phone:+81-3-5734-2955
06d19058@nr.titech.ac.jp

Hiroshi SEKIMOTO
Tokyo Institute of Technology
1-12-1 N1-17, Ookayama, Meguro-ku,
Tokyo
Phone:+81-3-5734-3066
hsekimoto@nr.titech.ac.jp

keywords: CANDLE, Fast reactor, Recladding

Abstract

A new reactor burnup strategy CANDLE was proposed, where shapes of neutron flux, nuclide densities and power density distributions remain constant but move to an axial direction.

Application of this burnup strategy to neutron rich fast reactors makes excellent performances. Only natural or depleted uranium is required for the replaced fresh fuels. About 40 % of natural or depleted uranium undergoes fission. But presently, there are no data for material integrity under a condition of 40% burnup.

In this paper, this we try to solve this problem by the following method. Fuels burning in progress are removed from core, and recladded. During this process FP gas is removed from these fuels. Then, they are charged in the previous position of core again. Finally, 40% burnup is attained with maintaining material integrity.

We performed the burn-up calculation including the above process and investigated reactor physics properties. As a result, attained maximum neutron fluence became small because of recladding. Effective neutron multiplication factor when reactor start up was 1.0015 and one just before reactor stopped was 1.003. The effects of recladding appeared small. Shape of each distribution was almost same and burnup was about 41%.

1. Introduction

A new reactor burnup strategy CANDLE (Constant Axial shape of Neutron flux, nuclide densities and power shape During Life of Energy producing) was proposed, where shapes of neutron flux, nuclide densities and power density distributions remain constant but move to an axial direction.

Application of this burnup strategy to neutron rich fast reactors makes excellent performances. Only natural or depleted uranium is required for the replaced fresh fuels. About 40 % of natural or depleted uranium undergoes fission. But presently, there are no data for material integrity, especially for cladding, under a condition of 40% burnup.
We try to solve this problem by the following method. Fuels burning in progress are removed from core, and recladding is performed. During this process, volatile FP gas is removed from these fuels. Then, they are charged in the previous position in the core again. Finally, 40% burnup is attained with maintaining material integrity. In this paper, we perform the burn-up calculation including the above process and investigate reactor physics properties.

2. Analysis method
2.1 In core fuel management

An example of in-core fuel management is shown for CANDLE reactor with recladding operation in Fig.1. Fuel elements “3a” and “6a” in Fig.1 are prepared beforehand for initial fuel exchange. These fuels (3a and 6a) contain similar composition as object fuels “4” and “7”. Then, fuel “4” and “7” are removed from core and fuel “3a” and “6a” are inserted in the corresponding positions, respectively. At the same time, fuel “1” that burned sufficiently is removed and new fuel “9” is added. After that, the reactor is restarted. During driving the reactor, recladding and removing volatile FP gas are performed for fuel “4” and “7”. Then, after driving one period, the reactor is stopped and the fuel exchange is performed again. At this time, fuel “5” and “8” are removed from core and fuel “4” and “7” are inserted. And fuel “2” is removed and new fuel “10” is added.

One of characteristics of CANDLE burnup strategy is that shapes of neutron flux, nuclide densities and power density distributions remain constant but move to an axial direction. So, operation period for one cycle is determined by moving velocity and height of removed fuel.

The calculation flow chart of this process is shown in Fig.2. After finishing burn-up calculation (left side (1) in Fig.2), object fuel data are updated (change fuels) and burn-up calculation is repeated. Recladding and decay calculation (right side(2) in...
Fig.2 Calculation flow chart

Fig.2) are performed for removed fuel data. Period of this decay calculation is the same as the period of next burnup of the core. The simulation is performed by repeating above process.

2.2 Burn-up calculation

The equation for diffusion calculation in burn-up calculation is basically as follows;

$$- \nabla D_g \nabla \phi_g + \sum_{i} \Sigma_{a,g} \phi_i + \sum_{g' \to g} \Sigma_{s,g \to g'} \phi_{g'} = \frac{1}{k_{efg}} \chi_g \sum_{g} \nu_g \sum_{f,g} \phi_f + \sum_{g' \to g} \chi_{g'} \phi_{g'}$$ (Eq.1)

where \( \phi_g \) is neutron flux in group “g”, \( D_g \) is diffusion factor in group “g”, \( \Sigma_{a,g} \) is absorption macro cross section in group “g”, \( \Sigma_{s,g \to g'} \) is scattering macro cross section from group “g” to group “g’”, \( \chi_g \) is fission macro cross section in group “g”, \( \chi_{g'} \) is fission spectrum, \( k_{efg} \) is effective neutron multiplication factor, and \( \nu_g \) is average number of fission neutrons emitted by fission reaction in group “g”.

The equation for burn-up calculation of burn-up calculation is basically as follows;

$$\frac{dN_j}{dt} = -\lambda_i N_j - \sum_g \sigma_{a,g,i} \phi_g N_j + \sum_f \lambda_j \alpha_{j \to i} N_j$$

$$+ \sum_{j \to g} \beta_{j \to g} \sigma_{a,g,j} \phi_g N_j + \sum_{j \to g} \gamma_{j \to g} \sigma_{f,g,j} \phi_g N_j$$ (Eq.2)

where \( \phi_g \) is neutron flux in group “g”, \( \lambda_i \) is decay constant of nuclide “i”, \( \sigma_{a,g,i} \) is an absorption micro cross section of nuclide “i” in group “g”, \( \sigma_{f,g,i} \) is a fission micro cross section of nuclide “i” in group “g”, \( \sigma_{a,g,j} \) is \( \sigma_{a,g,i} - \sigma_{f,g,i} \), \( \alpha_{j \to i} \) is a probability of decay from nuclide “j” to nuclide “i”, \( \beta_{j \to g} \) is a probability of nuclide “j” becoming nuclide “i” by absorbing neutron, and \( \gamma_{j \to g} \) is yield of fission product nuclide “i” by a fission of nuclide “j”.

2.3 Recladding

Recladding in this paper means that proper fuels, whose fast neutron dose become near maximum limit, are removed from the core, and their claddings are replaced by new ones and volatile FP gas is removed. How to calculate above process is as follows;

1. For removed fuels, decay calculation is performed until they are brought back to the core and burned there again. The equation of nuclide density change by decay is as follows;

$$\frac{dN_j}{dt} = -\lambda_i N_j + \sum_f \lambda_j \alpha_{j \to i} N_j$$ (Eq.3)

where \( \lambda_i \) is decay constant of nuclide “i”, \( N_j \) is number density of nuclide “i”, and \( \alpha_{j \to i} \) is probability of decay from nuclide “j” to nuclide “i”.

2. Regarding to removing volatile FP gas, krypton, xenon, and iodine, their number densities are set zero.
3. Regarding to changing cladding, number density of cladding that are damaged during burn-up is renewed to the value of new cladding.

2.4 Data for calculation

Neutron data for CANDLE code were obtained using SRAC code system\textsuperscript{[3]} with JENDL-3.3 nuclear data library. Fine 107 energy groups micro cross section set is calculated by SRAC and this is changed to coarse 21 group micro cross section set and used in the analysis.

The cell and core geometries \textsuperscript{[2]} of calculation model are shown in Fig.3 and Fig.4, respectively. Main design parameters are given in Table.1. Number density of each nuclide is given in Table.2.

![Fig.3 Cell geometry](image)

![Fig.4 Geometry of calculation model](image)

<table>
<thead>
<tr>
<th>Table.1 Main parameter and design</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core height[cm]</td>
</tr>
<tr>
<td>Fuel height[cm]</td>
</tr>
<tr>
<td>Fuel radius[cm]</td>
</tr>
<tr>
<td>Radial shield thickness[cm]</td>
</tr>
<tr>
<td>Outer cladding diameter [cm]</td>
</tr>
<tr>
<td>Cladding thickness[cm]</td>
</tr>
<tr>
<td>Fuel rod pitch[cm]</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Fuel</th>
<th>Natural uranium nitride (N15 enriched 99%,81%TD)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Cladding</td>
<td>HT-9</td>
</tr>
<tr>
<td>Coolant</td>
<td>Pb-Bi (44.5%-55.5%)</td>
</tr>
<tr>
<td>Power[MWth]</td>
<td>3000</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Table.2 Number density of components</th>
</tr>
</thead>
<tbody>
<tr>
<td>Nuclide</td>
</tr>
<tr>
<td>---------</td>
</tr>
<tr>
<td>Fuel</td>
</tr>
<tr>
<td>U235</td>
</tr>
<tr>
<td>U238</td>
</tr>
<tr>
<td>Cladding(HT9)</td>
</tr>
<tr>
<td>Fe54</td>
</tr>
<tr>
<td>Fe56</td>
</tr>
<tr>
<td>Fe57</td>
</tr>
<tr>
<td>Fe58</td>
</tr>
<tr>
<td>Cr50</td>
</tr>
<tr>
<td>Cr52</td>
</tr>
<tr>
<td>Cr53</td>
</tr>
<tr>
<td>Cr54</td>
</tr>
<tr>
<td>Nb93</td>
</tr>
<tr>
<td>Mo92</td>
</tr>
<tr>
<td>Mo94</td>
</tr>
<tr>
<td>Coolant</td>
</tr>
<tr>
<td>Pb</td>
</tr>
</tbody>
</table>

Copyright © 2007 by JSME
3. Result

If amount of neutron fluence is small, material integrity is high. So timing of recladding depends on amount of fast neutron fluence. In reference, irradiation limit is decided \(5.0 \times 10^{23} \text{ n/cm}^2\) \(^{[4]}\). I will show the result in case of 2 position recladding. Fast neutron fluence distributions at BOC for both with and without recladding are shown in Fig.5, where fuel blocks are removed and recladded. Positions of removed fuel blocks are 165 ~ 170cm and 225 ~ 240cm. In this analysis, a height of one fuel block is 15cm. In calculation, mesh size is 5cm in fuel region, so 3 meshes are regarded as one fuel block. By performing recladding neutron fluence becomes small and less than maximum permissible value.

![Fig.5 Fast neutron fluence distribution](image)

Fast fluence distributions at both beginning of cycle (BOC) and end of cycle (EOC) are shown in Fig.6. The operation period per 1 cycle is about 1620 days. It corresponds to the moving velocity of flux distribution of about 3.38 cm/year.

![Fig.6 Fast neutron fluence distributions of beginning of cycle (BOC) and end of cycle (EOC)](image)

Infinite neutron multiplication factors of BOC and EOC are shown in Fig.7. The distribution moves from right to left. These distribution curves are not smooth. It may be attributed to two reasons. One is that neutron economy of the recladded fuel becomes better by removing volatile FP gas. The other is the radioactive decay of nuclei, especially \(^{241}\text{Pu}\) during out of the core. They affect the characteristics of the core. Half-lives of \(^{238}\text{U}\), \(^{239}\text{Pu}\), and \(^{240}\text{Pu}\) are \(4.47 \times 10^9\) years, \(2.41 \times 10^4\) years, and \(6.57 \times 10^3\) years, respectively. But the half-life of \(^{241}\text{Pu}\) is 14.335 years, so this decay effect cannot be ignored.

![Fig.7 Infinite neutron multiplication factor](image)

The atomic number densities of \(^{241}\text{Pu}\) for both with and without recladding are shown in Fig.8. The parts of 165 ~ 170cm and 225 ~ 240 cm in BOC are dented,
since $^{241}\text{Pu}$ decays while removed fuel blocks are out of the core.

Effective neutron multiplication factors for both with and without recladding are shown in Fig.9. In this figure, effective neutron multiplication factor is calculated for 6 cycles. The operation and stopping period for each cycle are 1620 days and 100 days, respectively. The effective neutron multiplication factor is about 1.0015 at BOC and about 1.003 at EOC.

Power distributions at BOC and EOC for both with and without recladding are shown in Fig.10. Both shapes are almost the same shape. Between BOC and EOC, the shape does not change with burn-up.

4. Conclusions

For maintaining material integrity, recladding was adopted in CANDLE fast reactor and exposed neutron fluence can be kept under the limit, even though the 41% burnup of discharged fuel is attained. The effects on reactor physics characteristics of recladding appeared small. This method can keep characteristics of CANDLE burnup strategy.

In the present paper we have not discussed so much on the fuel structure and actual refueling procedure. These problems will be studied in the future works.

5. Reference

